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FROM
INDUCED GAMMA ACTIVITY
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POWER LEVEL DETERMINATION FROM INDUCED GAMMA ACTIVITY IN NUCLEAR REACTOR COOLANTS

by

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PREFACE

The operation of a nuclear power plant involves a variety of instrumentation, most of which is of a conventional character. However, for control purposes, conventional instrumentation for power level determination is not adequate since it does not respond instantaneously either to overall or to local changes in reactor power. The reactor power level is proportional to the neutron flux, and instruments that measure flux can be located at many points throughout the reactor system. Such devices have an essentially instantaneous response and are suitable for indicating power levels and for operating automatic control and safety mechanisms.

With respect to power level determination, the Westinghouse Corporation, through operating experience with the prototype of the Submarine Thermal Reactor, has determined that there are two basic deficiencies in the neutron detection systems. These are that (a) an array of detection instruments is required in order to minimize the effect of changes in control rod configuration, thus complicating the instrumentation by requiring either multiple indicators or some sort of scanning-averaging circuit, and (b) there is an aging phenomenon associated with neutron detection instruments when used in high flux regions for extended periods, thus requiring periodic calibrations with thermal instrumentation and eventually replacement.

It would be desirable to have a system for power level determination which incorporates the features of the neutron detection systems, without these deficiencies. It is the aim of this paper to describe one such system, and evaluate its probable behavior under different operating conditions.

In order to avoid any breach of security, the writer has not used any actual dimensions or characteristics which would apply specifically to the Submarine Thermal Reactor. However, an attempt has been made to arrive at results which are close enough to the actual case to have some meaning when compared with the experimental results. For this same reason, any description of actual systems is either in very general terms or left to the reader to look up in the classified material available.

Experimental work in connection with this paper was done at the Submarine Thermal Reactor Test Site, Arco, Idaho, with the assistance and cooperation of the employees of the Westinghouse Corporation at that facility. The writer particularly wishes to thank Doctor W. H. Esselman, Head of the Technical Section, who presented the original idea for investigation, and Mr. Grady B. Matheny of that section who served as a technical advisor during the experimental work. Also, Professor E. E. Drucker of the U. S. Naval Postgraduate School, for his assistance, encouragement, and cooperation in the preparation of this paper.

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CHAPTER I

INTRODUCTION

1. If N radioactive nuclei are present at time t, and if no new nuclei are introduced into the sample, the rate of change of active nuclei with respect to time is

$$\frac{dN}{dt} = -\lambda N \tag{1}$$

$$\therefore \quad N = N_0 e^{-\lambda t} \tag{2}$$

where λ = the disintegration constant = $\frac{0.693}{T}$ (3)

$\frac{dN}{dt}$ or something proportional to $\frac{dN}{dt}$ is observed rather than N.

Differentiating (2) and substituting from (1)

$$\frac{dN}{dt} = \left(\frac{dN}{dt}\right)_0 e^{-\lambda t}$$

or

$$R = R_0 e^{-\lambda t} \tag{4}$$

The disintegration rate R is called the activity of the sample.

2. In chain decay equation (2) holds for the first member. Let species A decay into species B with disintegration constant λ_A; Let B decay into C with disintegration constant λ_B; one does not care what happens to C. Let N_A and N_B be the number of A and B atoms present at time t. For N_A one has from equation (2)

$$N_A = N_{A,0} e^{-\lambda_A t} \tag{5}$$

The net rate at which N_B is changing $\left(\frac{dN_B}{dt}\right)$ is the difference between the rate at which it is being produced $\left(-\frac{dN_A}{dt}\right)$ and the rate of which it is decaying $[\lambda_B N_B$; See equation (1)$]$.

Using equation (5) this gives

$$\frac{dN_B}{dt} = \lambda_A N_{A,0} e^{-\lambda_A t} - \lambda_B N_B \tag{6}$$

Solving this differential equation for N_B and assuming that one starts with $N_B = 0$ at $t=0$ leads to

$$N_B = \frac{\lambda_A}{\lambda_B - \lambda_A} N_{A,0} \left(e^{-\lambda_A t} - e^{-\lambda_B t}\right) \tag{7}$$

3. A limiting case of equation (7) is that in which A is very long-lived or $\lambda_A \ll \lambda_B$. At large t, $e^{-\lambda_B t}$ can always be neglected relative to $e^{-\lambda_A t}$. N_B then approaches

$$N_B = N_{A,0}\, e^{-\lambda_A t} \left(\frac{\lambda_A}{\lambda_B}\right) \tag{8}$$

If the period of A is not only much larger than that of B but also is large compared with the time during which one makes observations, $e^{-\lambda_A t}$ will not change much during the observation, and N_B will approach a constant "secular equilibrium" value $N_{B\infty}$ given by

$$N_{B\infty} = N_A \left(\frac{\lambda_A}{\lambda_B}\right) \tag{9}$$

This same value of $N_{B\infty}$ is approached no matter what the initial value of N_B is.

Since $\lambda_A \ll \lambda_B$ with $\lambda_A \approx 0$, and substituting from equation (9), equation (7) may now be written

$$N_B = N_{B\infty} \left(1 - e^{-\lambda_B t}\right) \tag{10}$$

Secular equilibrium occurs when a radioactive substance is being made at a constant rate either by the decay of its long-lived parent (as above) or by bombardment in a cyclatron or nuclear reactor. In either event, after waiting approximately six half-lives of B, N_B would approach its equilibrium value, where the decay rate ($N_{B\infty} \lambda_B$) equals the constant production rate ($N_A \lambda_A$); See equation (9).

4. For the particular case under consideration (i.e. neutron reactions with the coolant in a reactor producing a gamma radioactive species) the constant production rate is $\Sigma\phi$, where Σ is the macroscopic cross-section of the coolant and ϕ is the flux within the reactor core (which will be assumed uniform and monoenergetic for purposes of this paper). Sub-

stituting this value for $N_A \lambda_A$ in equation (9) leads to the secular equilibrium value given by

$$N_{B\infty} = \frac{\Sigma \phi}{\lambda_B}$$

$$\text{or } N_{B\infty} \lambda_B = \Sigma \phi \qquad (11)$$

If t = 0 is considered to be the time a unit volume of coolant leaves the region of high flux; the instantaneous initial decay rate, by substitution of equation (11) in equation (1), is found to be

$$\left(-\frac{dN_B}{dt}\right)_0 = N_{B\infty} \lambda_B = \Sigma \phi \qquad (12)$$

At a time t after leaving the high flux region the instantaneous decay rate will be, from equation (4)

$$\frac{dN_B}{dt} = \left(\frac{dN_B}{dt}\right)_0 e^{-\lambda_B t}$$

$$\text{or } R = R_0 e^{-\lambda_B t} \qquad (13)$$

Where R_0 is proportional to $\Sigma \phi$.

$$(\text{i.e. } R_0 = K' \Sigma \phi \qquad) \qquad (14)$$

It may occur to the reader that the unit volume of coolant leaves the reactor core, traverses the loop piping, then returns to the high flux region, and therefore may not actually reach the secular equilibrium condition before leaving the reactor again. This is not the case, as the relative magnitudes of the times involved (i.e. loop transit time, time within the core, and half life) are such as to minimize this effect. This will be discussed further in the next chapter using the system described there for examples.

5. The power level of a given power reactor is obviously directly related to the rate of heat release within the core. The rate of heat release is dependent on the fission rate. The fission rate for the reactor is $\Sigma_{\text{fuel}} \phi$, therefore

$$\text{Power} = K'' \Sigma_{fuel} \phi \qquad (15)$$

Since for a given reactor installation K" and Σ_{fuel} will be constant, power is directly proportional to flux (or vice versa).

Substituting equations (14) and (15) into equation (13) gives

$$R = R_o e^{-\lambda_B t} = (K'\Sigma\phi) e^{-\lambda_B t} = \left[K'\Sigma\left(\frac{Power}{K''\Sigma_{fuel}}\right)\right] e^{-\lambda_B t}$$

$$\text{or } R = KP e^{-\lambda_B t} \qquad (16)$$

where $K = \dfrac{K'\Sigma}{K''\Sigma_{fuel}}$

and $P = \dfrac{\text{Percent Power}}{100}$

6. The time, t, in equation (16) will be the time required for a given unit volume to travel from the reactor core outlet to a fixed point of measurement.

Let A = Flow Area of Coolant Loop Piping

F = Fraction of Full Flow (as a decimal)

L = Length of Piping from Core Outlet to Point of Measurement

Q = Capacity of Coolant System

T = Time for a Unit Volume to Complete the Flow Cycle at Full Flow

then

$$t = \frac{TAL}{QF} \qquad (17)$$

Substituting in equation (15)

$$R = KP\, e^{-\lambda_B\left(\frac{TAL}{QF}\right)} \qquad (18)$$

for a given system this may be written as

$$R = KP\, e^{-\frac{C}{F}} \qquad (19)$$

where $C = \dfrac{\lambda_B TAL}{Q}$

(j) $N^{16}(n, \gamma) N^{17}$ - N^{17} emits a 3.7 Mev beta in going to O^{17}, then a 0.9 Mev neutron and ends as O^{16}.

3. It can be seen from the preceeding section that there are only four possible reactions which should be considered. These are the $O^{16}(n, p)$ N^{16}, $O^{17}(n, p) N^{17}$, $O^{18}(n, \gamma) O^{19}$, and the $N^{16}(n, \gamma) N^{17}$. Using Feather's formula for a rough range-energy relation for the beta radiation [4]

$$\text{range } (g/cm^2) = 0.546E(\text{Mev}) - 0.016 \qquad (1)$$

one finds the range of the most energetic beta emitted, from N^{16}, to be

$$\text{range} = (0.546)(10) - 0.016 \doteq 5.30 \ g/cm^2. \qquad (2)$$

With an average density of steel, used for reactor structure and piping assumed as 7.9 g/cm^3, this leads to a thickness of steel of less than one cm. ($\frac{5.3}{7.9} = 0.67$) to reduce this radiation to a point where it is indistinguishable from the background. It is therefore reasonable to assume that any beta activity can be neglected.

The threshold for the O^{17} (n,p)N^{17} reaction, which results in the emission of the 0.9 Mev neutron, is 7.9 Mev. The average cross section above the threshold is reported as 0.01 barn for an average neutron spectrum well above the threshold energy, but is not considered more reliable than by about a factor of 10. The natural abundance of O^{17} is 0.039%. Considering the abundance, reaction cross-section, and the attenuation probability, it is reasonable to assume that any effect from this neutron emission will be negligible.

The isotope O^{18} has an abundance of 0.20% and a thermal capture cross-section of 2.2 x 10^{-4} barn. The O^{18} $(n, \gamma)O^{19}$ reaction emits a 1.6 Mev delayed gamma with a 29 second half-life. Comparing abundance,

reaction cross-section, and average flux values, the equilibrium activity due to this reaction will be roughly 10% of the N^{16} activity. Since in addition the delayed gamma from O^{19} is far less penetrating than that from N^{16}, it is reasonable to assume that the escaping O^{19} delayed gamma activity will be negligible.

4. As a result of the above process of elimination, there remains only one induced activity in the coolant which need be considered; the N^{16} delayed gamma which results from the O^{16} (n,p) N^{16} reaction. The threshold for this reaction is 9.5 Mev. The reaction cross-section, averaged over the fission spectrum is 0.014×10^{-3} barn. The half life of the N^{16} product is 7.35 seconds, and the gamma energy is 6 Mev. Seventy five percent of the N^{16} delayed gamma decays are accompanied by this high energy gamma.

5. Based on the assumptions of articles 3 and 4, above, a rough and simple system was devised. The system consisted of

(a) The Detector- An anthracene crystal and a 5819 photomultiplier tube

(b) The Indicator- An Electrometer, Model 4-565-1, Mfgr. Victoreen Instrument Company.

The detector was heavily shielded to prevent any interference from stray radiations, and to give it directional properties. A check of the angularity at the detector, using a 77 mc Co^{60} source at a distance of one foot, indicated a beam width of approximately 9.54 degrees.

The output of the detector was put to the Electrometer and null voltage readings used for plotting purposes. Later in the experimental work the detector output was fed to a Sanborn Recorder for continuous readings, and occasional null voltage readings were taken to correlate

the Sanborn scale with the Electrometer readings for the previous runs.

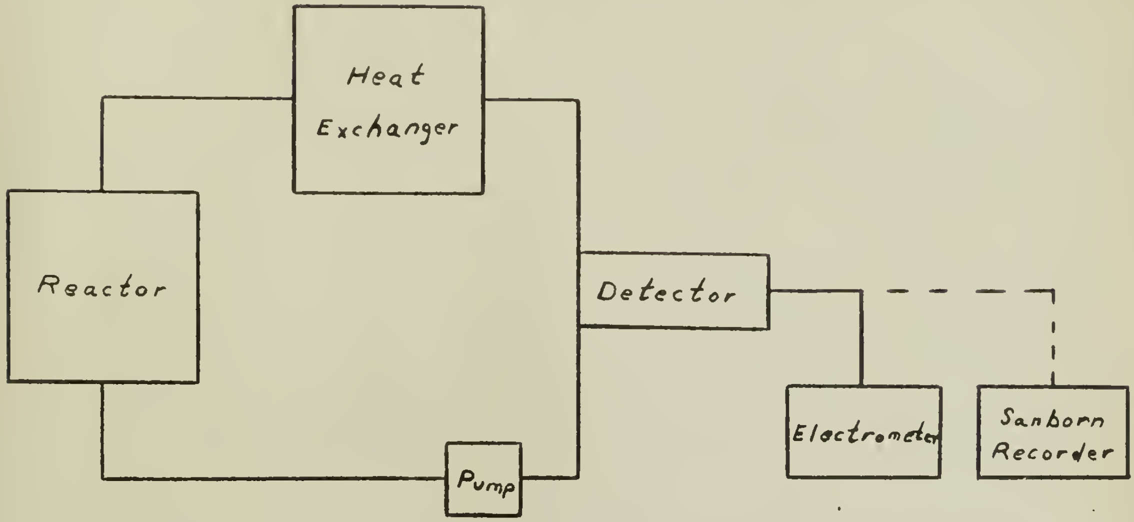


Fig. 1 - Simplified Block Diagram of Experimental System Showing Relative Positions of Equipment

With the physical layout of the system as shown above, and inasmuch as the only induced reaction of concern is the O^{16} (n,p) N^{16} reaction, the following constants may be assumed to hold for use in equation (19) of Chapter I:

A = 1.70 Square Feet

L = 26 Feet

Q = 400 Cubic Feet

T = 20 Seconds

$$\lambda = \frac{0.693}{7.35} = 0.0943/\text{Seconds}$$

Equation (18) of Chapter I may now be written

$$R = KP\, e^{-\frac{(0.0943)(20)(1.70)(26)}{400F}}$$

$$\text{or } R = KP\, e^{-\frac{0.2085}{F}} \qquad (3)$$

In this form the equation will apply to the particular experimental system used.

6. At this point, it is advisable to stop and consider the effect of flow on the equilibrium concentration of N^{16}.

From equation (10) of Chapter I

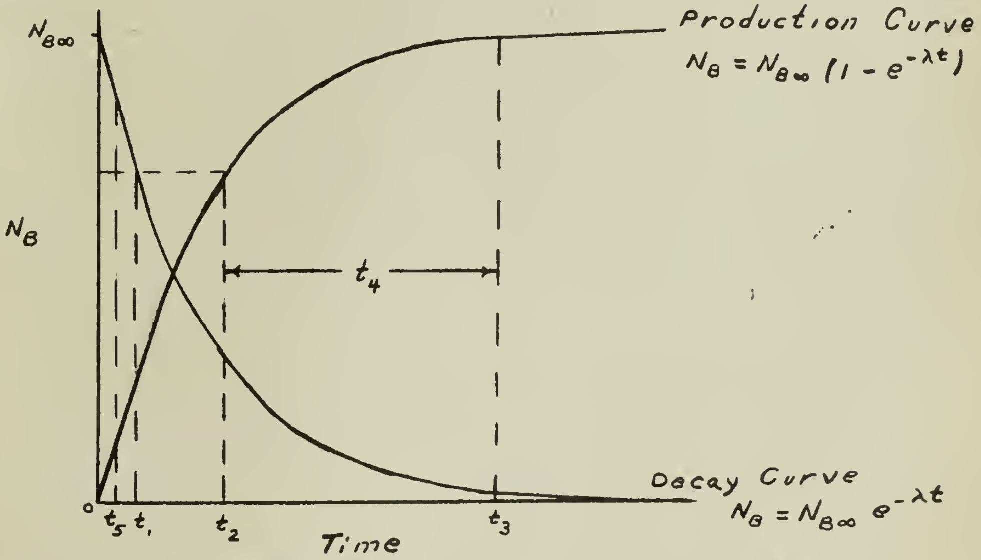
$$N_B = N_{B\infty} \quad (1-e^{-\lambda_B t}) \tag{4}$$

where t = 0 at the effective time of entering the core. The number of activated N^{16} nuclei in the coolant, at any time t while in the core, may be found from equation (4) above.

Assuming that equilibrium is attained in passing through the core, the number of activated nuclei in the coolant, at any time t after leaving the core, may be obtained from the simple decay law (Equation (2) of Chapter I) as

$$N_B = N_{B\infty} \quad e^{-\lambda_B t} \tag{5}$$

To determine the effect of flow on the equilibrium value of N^{16}, equations (4) and (5), above, must be considered simultaneously in the following manner:



Let t_1 = loop time (or decay time before reentering the reactor)

t_2 = Effective initial production time (or time which would have been needed to build up to same value of N_B, along normal production curve, as that with which the coolant is reentering the reactor)

t_3 = Effective total production time = $t_2 + t_4$

t_4 = Actual time in core (actually a Δt)

t_5 = Time from reactor outlet to detector

At equilibrium; t_1 , t_2 , and t_3 must be so related that

$$N_{B\infty}\,(1-e^{-\lambda t_2}) = N_{B\infty}\,(1-e^{-\lambda t_3})\, e^{-\lambda t_1} \tag{6}$$

Utilizing the fact that t_1 and t_4 can be expressed as "known" functions of flow and that $t_3 = t_2 + t_4$, solving equation (6) for $e^{-\lambda t_2}$ one obtains

$$e^{-\lambda t_2} = \left[\frac{1-e^{-\lambda t_1}}{1-e^{-\lambda(t_1+t_4)}}\right] \tag{7}$$

Since

$$e^{-\lambda t_3} = e^{-\lambda(t_2+t_4)} = e^{-\lambda t_2}\quad e^{-\lambda t_4} \tag{8}$$

then

$$e^{-\lambda t_3} = \left[\frac{1-e^{-\lambda t_1}}{1-e^{-\lambda(t_1+t_4)}}\right] \quad e^{-\lambda t_4} \tag{9}$$

and

$$1-e^{-\lambda t_3} = \left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right] \tag{10}$$

Therefore at equilibrium, the number of activated nuclei per unit volume of coolant at the reactor outlet

$$N_B = N_{B\infty}\,(1-e^{-\lambda t_3}) = N_{B\infty}\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right] \tag{11}$$

This indicates that equation (3) of this chapter should be modified to include a factor for the effect of flow on the equilibrium activity within the coolant. This equation then takes the form

$$R = KP\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right] e^{-\frac{C}{F}} \tag{12}$$

Substituting the constants from article 5, Chapter II into equation (12)

$$t_1 = \frac{TAL'}{QF} = \frac{0.272}{F} \qquad \therefore\ \lambda t_4 = \frac{1.63}{F} \tag{13}$$

$$t_1 + t_4 = \frac{T}{F} = \frac{20}{F} \qquad \therefore\ \lambda(t_1+t_4) = \frac{1.89}{F} \tag{14}$$

$C = 0.2085$

and

$$R = KP\left[\frac{1 - e^{-\frac{1.63}{F}}}{1 - e^{-\frac{1.89}{F}}}\right] e^{-\frac{0.2085}{F}} \qquad (15)$$

The total effect of flow on the activity indicated by the detection system, is best indicated by the following table.

Table 1 - Correction Factor for Flow Effect

F	$\left[\frac{1-e^{-\frac{1.63}{F}}}{1-e^{-\frac{1.89}{F}}}\right]$	$e^{-\frac{0.2085}{F}}$	Correction Factor	Percent Variation of Factor from Factor for $F = 1.00$
1.00	0.947	0.812	0.769	----
0.95	0.950	0.803	0.763	0.8
0.90	0.953	0.793	0.755	1.8
0.85	0.957	0.782	0.749	2.6
0.80	0.961	0.770	0.740	3.8
0.75	0.964	0.757	0.722	6.1
0.70	0.968	0.742	0.718	6.6
0.65	0.971	0.725	0.704	8.5
0.60	0.976	0.706	0.689	10.4
0.55	0.981	0.685	0.672	12.1
0.50	0.984	0.659	0.649	15.6

Inasmuch as the dimensions and times used for calculations are only approximations, and estimates based on these approximations, the variations as a result of flow changes are not prohibitive. The actual dimensions and times involved in the STR power plant are such as to reduce this variation.

CHAPTER III

EXPERIMENTAL RESULTS

1. In Chapters I and II it has been shown that the induced activity in the reactor coolant is proportional to the power level of the reactor, and that it is possible to devise an instrument system which uses this activity as a means of power level determination. A system such as described in article 5, Chapter II, was installed at the Submarine Thermal Reactor Test Site and the following data obtained.

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER ONE					
0.0	17.2	52.0	——	0.0	0.0
1.5	17.2	60.3	——	8.3	8.3
2.6	17.2	69.6	——	17.6	17.6
8.5	17.2	76.5	——	24.5	24.5
9.5	17.2	85.0	——	33.0	33.0
10.75	17.2	93.0	——	41.0	41.0
11.5	17.2	93.0	——	41.0	41.0
13.0	17.2	97.8	——	45.8	45.8
RUN NUMBER TWO					
12.5	96.5	101.0	——	53.0	56.5
15.0	96.5	119.0	——	61.0	65.0
16.0	96.5	122.0	——	64.0	68.0
18.0	96.5	129.0	——	71.0	75.5
21.5	96.5	143.0	——	85.0	90.5

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER TWO (CONT.)					
25.0	96.5	155.4	——	97.4	104.0
27.5	96.5	155.0	——	97.0	103.0
30.0	96.5	157.0	——	99.0	105.0
33.0	96.5	172.7	——	114.7	122.0
40.0	96.5	189.3	——	131.3	140.0
41.0	96.5	192.8	——	134.8	144.0
41.5	96.5	192.5	——	134.5	143.0
42.0	96.5	197.0	——	139.0	148.0
42.5	96.5	196.3	——	138.3	147.0
44.0	96.5	202.0	——	144.0	153.0
45.0	96.5	207.7	——	149.7	159.0
46.0	96.5	209.0	——	151.0	161.0
47.0	96.5	211.0	——	153.0	163.0
47.5	96.5	215.5	——	157.5	168.0
48.0	96.5	222.8	——	164.8	175.0
49.0	96.5	224.4	——	166.4	177.0
49.5	96.5	222.0	——	164.0	175.0
50.0	96.5	226.0	——	168.0	179.0
50.5	96.5	229.0	——	171.0	182.0
51.0	96.5	225.0	——	167.0	178.0
52.0	96.5	231.0	——	173.0	184.0
52.5	96.5	231.0	——	173.0	184.0
54.0	96.5	231.0	——	173.0	184.0

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER TWO (CONT.)					
55.0	96.5	235.0	——	177.0	188.0
56.0	96.5	245.0	——	187.0	199.0
57.5	96.5	247.0	——	189.0	201.0
58.0	96.5	253.0	——	195.0	208.0
58.5	96.5	255.0	——	197.0	210.0
59.0	96.5	259.0	——	201.0	214.0
60.0	96.5	261.0	——	203.0	216.0
61.0	96.5	263.0	——	205.0	218.0
62.0	96.5	265.0	——	207.0	220.0
62.5	96.5	268.5	——	210.5	224.0
63.0	96.5	267.0	——	209.0	222.0
65.0	96.5	279.0	——	221.0	235.0
67.5	96.5	285.0	——	227.0	242.0
68.0	96.5	291.0	——	233.0	248.0
69.0	96.5	277.0	——	219.0	233.0
70.0	96.5	293.5	——	235.5	245.0
71.0	96.5	294.3	——	236.3	252.0
71.5	96.5	292.8	——	234.8	250.0
72.0	96.5	294.0	——	236.0	251.0
72.5	96.5	301.0	——	243.0	258.0
74.5	96.5	308.0	——	250.0	266.0
76.0	96.5	308.0	——	250.0	266.0
76.5	96.5	310.0	——	252.0	268.0

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER THREE					
37.5	80.0	187.0	——	135.0	133.0
38.0	80.0	207.0	——	155.0	152.0
40.0	80.0	207.0	——	155.0	152.0
47.5	80.0	227.0	——	175.0	172.0
67.5	80.0	295.0	——	243.0	239.0
69.0	80.0	301.0	——	249.0	245.0
RUN NUMBER FOUR					
5.0	96.5	51.0	370	17.0	18.0
13.5	96.5	84.0	620	50.0	53.0
25.0	96.5	113.0	850	79.0	84.0
58.0	96.5	212.0	1540	178.0	190.0
58.5	96.5	212.0	1540	178.0	190.0
59.0	96.5	210.0	1560	176.0	188.0
61.0	96.5	237.0	1760	203.0	216.0
RUN NUMBER FIVE					
5.0	96.5	57.0	410	13.0	17.5
10.0	96.5	73.0	530	29.0	39.0
15.0	96.5	88.0	640	44.0	59.5
20.0	96.5	101.0	740	67.0	90.5
25.0	96.5	115.0	840	71.0	96.0
30.0	96.5	124.0	910	80.0	108.0
35.0	96.5	139.0	1020	95.0	128.0
40.0	96.5	143.0	1050	99.0	134.0

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER FIVE (CONT.)					
45.0	96.5	156.0	1140	112.0	152.0
50.0	96.5	170.0	1250	126.0	170.0
55.0	96.5	183.0	1350	139.0	188.0
60.0	96.5	201.0	1480	157.0	212.0
RUN NUMBER SIX					
8.0	92.5	70.0	490	20.0	28.0
10.0	96.5	74.0	520	24.0	34.0
10.5	96.5	73.0	510	23.0	32.5
15.0	96.5	81.0	570	31.0	44.0
20.0	96.5	100.0	710	50.0	70.5
25.0	96.5	110.0	790	60.0	84.5
29.0	96.5	122.0	890	72.0	102.0
30.0	96.5	125.0	900	75.0	106.0
34.0	96.5	138.0	990	88.0	124.0
35.0	96.5	138.0	990	88.0	124.0
40.0	96.5	144.0	1040	94.0	133.0
44.0	96.5	161.0	1170	111.0	157.0
45.0	96.5	169.0	1220	119.0	168.0
48.0	96.5	177.0	1270	127.0	179.0
49.5	96.5	174.0	1200	124.0	175.0

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER SEVEN					
5.5	96.5	52.0	310	14.0	18.5
10.0	96.5	60.0	350	22.0	29.5
15.0	96.5	75.0	460	37.0	49.5
20.0	96.5	86.0	530	48.0	64.0
25.0	96.5	104.0	650	66.0	88.0
30.0	96.5	118.0	740	80.0	107.0
34.5	96.5	127.0	790	85.0	114.0
35.0	96.5	127.0	800	85.0	114.0
39.5	96.5	144.0	910	106.0	142.0
40.0	96.5	144.0	910	106.0	142.0
41.0	96.5	152.0	970	114.0	152.0
44.0	96.5	155.0	980	117.0	156.0
45.0	96.5	155.0	980	117.0	156.0
49.0	96.5	168.0	1060	130.0	174.0
49.5	96.5	163.0	1030	125.0	167.0
50.0	96.5	162.0	1030	124.0	166.0
50.5	96.5	175.0	1100	137.0	183.0
55.0	96.5	166.0	1050	128.0	171.0
60.0	96.5	173.0	1100	135.0	181.0
65.0	96.5	209.0	1320	171.0	229.0
67.5	96.5	218.0	1400	180.0	241.0
70.0	96.5	223.0	1420	185.0	248.0
75.0	96.5	236.0	1500	198.0	265.0

% Power	% Flow	Reading	"Scale"	Reading Corrected for Background	Equivalent Reading
RUN NUMBER EIGHT					
40.0	96.5	145.0	860	47.5	116.0
45.0	96.5	163.0	970	65.5	160.0
50.0	96.5	170.0	1010	72.5	177.0
55.0	96.5	176.0	1050	78.5	192.0
60.0	96.5	184.0	1100	86.5	212.0
65.0	96.5	191.0	1140	93.5	228.0
RUN NUMBER NINE					
70.0	96.5	203.0	1220	228.0	249.0
75.0	96.5	220.0	1320	245.0	266.0
80.0	96.5	233.0	1400	258.0	280.0
85.0	96.5	246.0	1480	271.0	294.0
90.0	96.0	272.0	1640	297.0	322.0
95.0	96.5	290.0	1735	315.0	342.0
100.0	96.5	301.0	1815	326.0	354.0

2. A plot of the actual reading versus percent power (Figures Two and Three) indicates that the induced activity in the coolant is, indeed, linear with respect to power. However, there is an obvious variation in the slope of the line for each run; and a different background constant for each run.

As each line can be represented by the equation $R_n'' = m_n P + C_n$ the equations for the various lines are:

Run Number One; $R_1'' = 354\,P + 58$

Run Number Two; $R_2'' = 333\,P + 52$

Run Number Three; $R_3'' = 360P + 52$

Run Number Four; $R_4'' = 332P + 34$

Run Number Five; $R_5'' = 262P + 44$

Run Number Six; $R_6'' = 251P + 50$

Run Number Seven; $R_7'' = 265P + 38$

Run Number Eight; $R_8'' = 145P + 97.5$

Run Number Nine; $R_9'' = 326.5P-25$

In order to compare the various runs, each run must be corrected for background, and then corrected for the differences in slopes. Converting all data points to equivalent readings, which will correspond to run number one, involves the following mathematical processes with each data point:

R_n' = Reading Corrected for Background = $R_n'' - C_n$

R_n = Equivalent Reading = $\frac{m_1}{m_n} R_n'$

The results of this manipulation are tabulated in the data tables, and are plotted in Figure Four. To briefly describe Figure Four, the resulting relationship between the equivalent reading and percent power is linear. Of the 135 experimental points, 102 (or 75.6% of the total) vary by four percent or less from the value which would be predicted from this line. Of this 102, it is interesting to note, that 40 (or 29.6% of the total) vary by approximately zero percent. It should also be noted that the data has not been corrected for flow effect, thus indicating that the effect of flow is not as great as that determined theoretically in Chapter II.

It may be reasoned that the variations in the slopes of the plots of original readings are due to impurities and flow rather than any effect of control rod configuration as:

(a) there was little change in the control rod configuration during the experimental period; and

(b) correcting the experimental readings for background, plotting these, and extrapolating all runs to zero power will result in all runs intersecting at a common point at zero power.

Theoretically any radial effect of control rod configuration will be well masked and eliminated by mixing in the header prior to entering the external loop piping. Changes in control rod configuration may have some influence on the coolant activity at the reactor outlet due to axial displacement of the point of maximum flux within the core. However, no calculations can be made on this effect without going into the actual construction and loading of the core. Since, for normal operation, control rod configuration should be such as to give as uniform a power and flux distribution as possible, the flux peak will not be too pronounced and will not drift much during the normal operating period between thermal calibrations. This effect may be such as to either increase or decrease the apparent power level depending on the reactor design. It is estimated that, over the normal operating life of the core, this effect will be of such small magnitude as to be easily accounted for during periodic thermal calibrations of the entire instrument system.

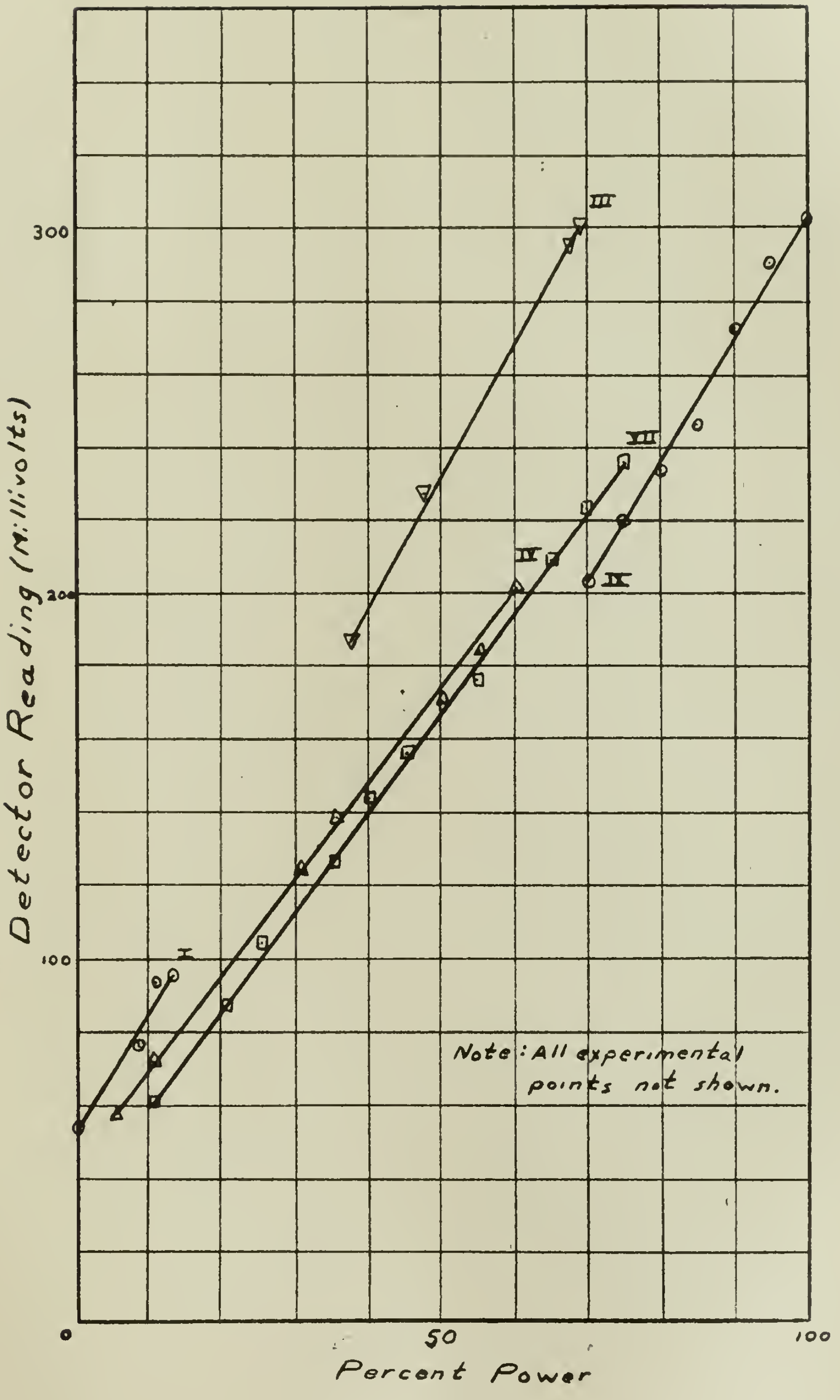
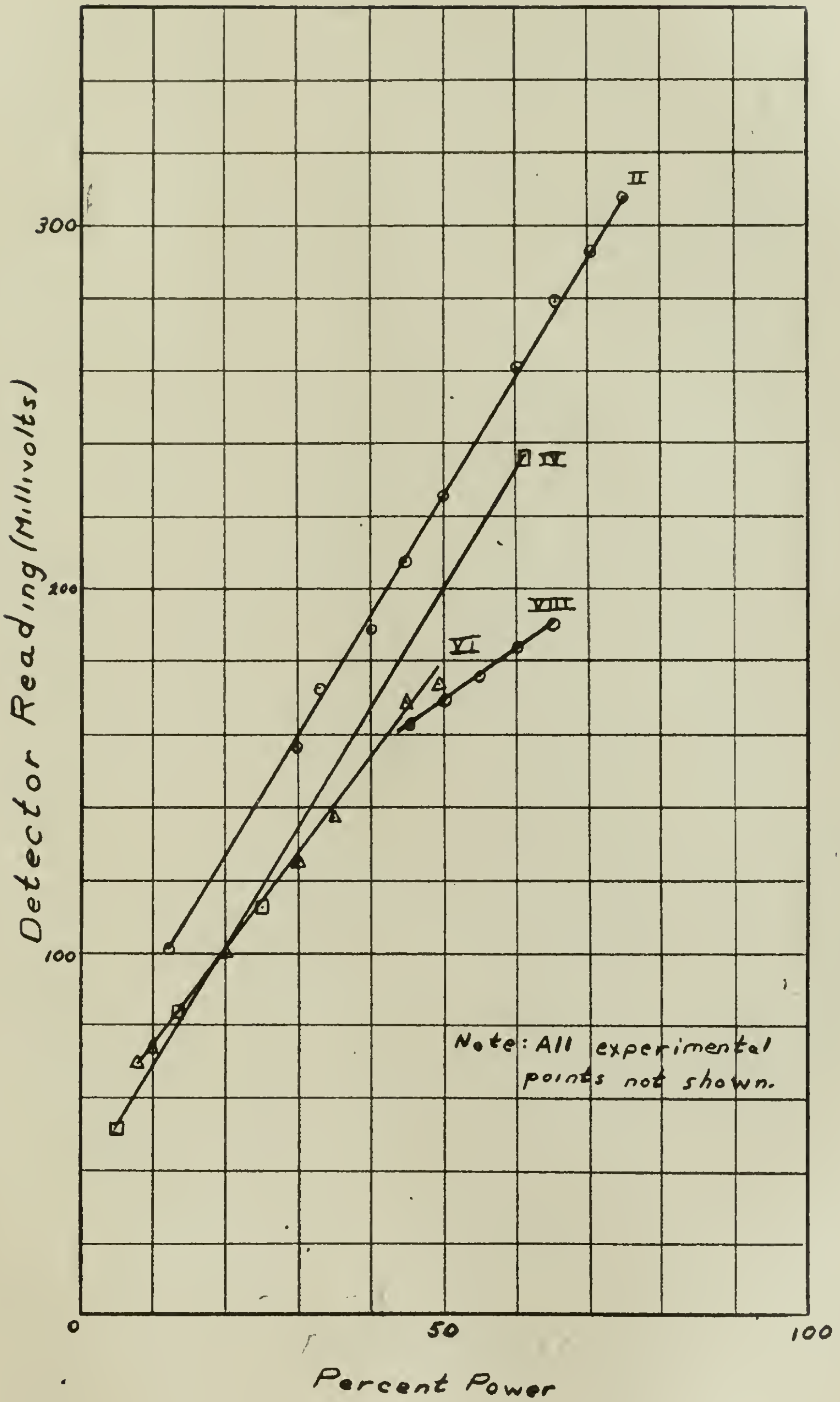
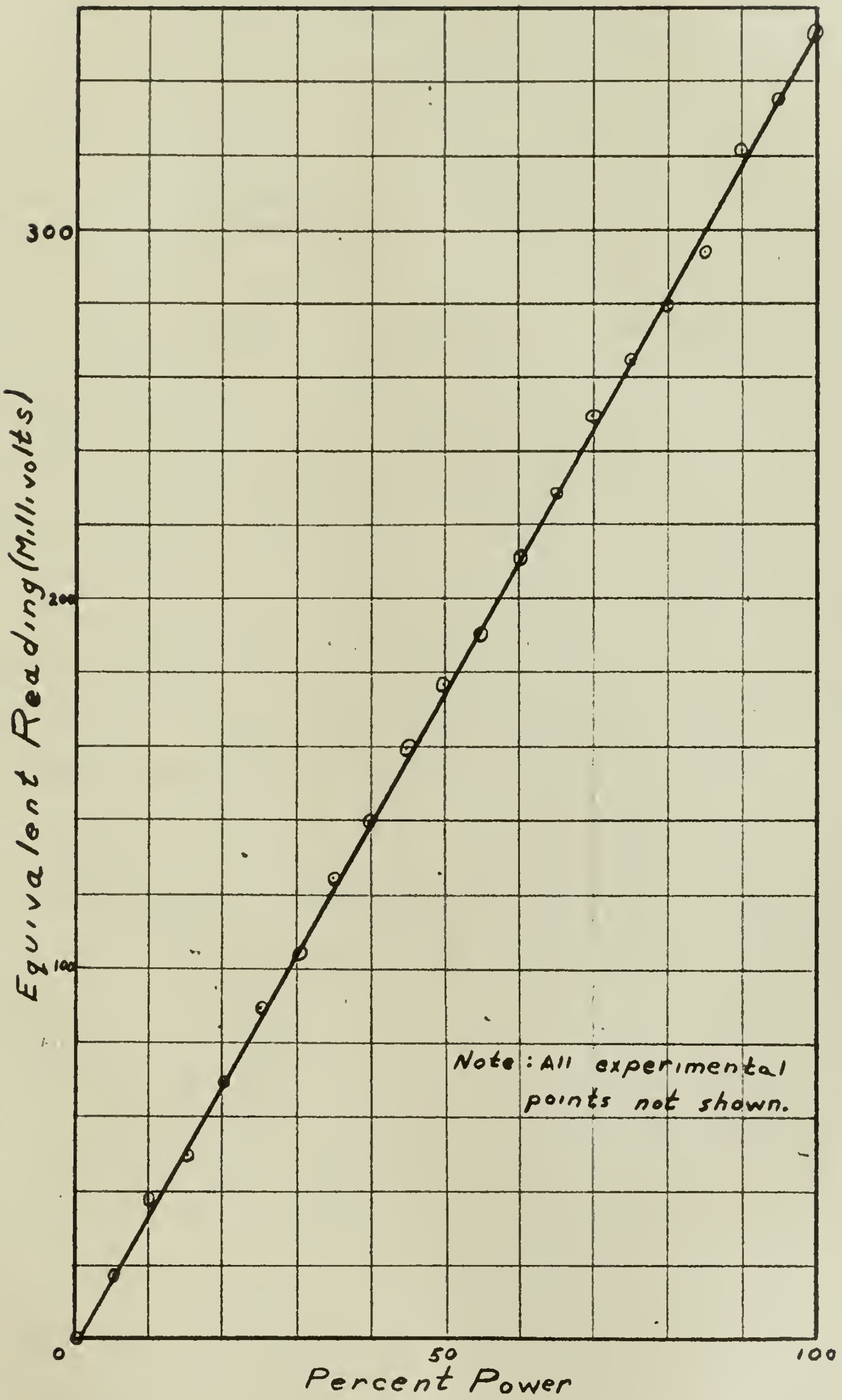

Detector Reading (Millivolts)
Percent Power
0
50
100
100
200
300
I
III
IV
VII
IX
Note: All experimental points not shown.

Figure 2



Note: All experimental points not shown.

Figure 3



Note: All experimental points not shown.

Figure 4

CHAPTER FOUR

EVALUATION OF SYSTEM

1. It has been shown that the number of activated nuclei per unit volume of coolant, at the primary coolant outlet of a given reactor system, can be described by the equation

$$N_{B,0} = N_{B\infty}\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right]$$

$$\text{or } N_{B,0} = \frac{\Sigma\phi_{av}}{\lambda}\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right] \qquad (1)$$

From the simple radioactive decay law, the decay rate of this unit volume at any time t_5 after leaving the reactor is

$$\frac{dN_B}{dt} = -\lambda N_B = -\Sigma\phi_{av}\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right]e^{-\lambda t_5} \qquad (2)$$

As flux is directly proportional to power, this equation can be written

$$\frac{dN_B}{dt} = K\,\frac{\Sigma_{coolant}}{\Sigma_{fuel}}\,P\left[\frac{1-e^{-\lambda t_4}}{1-e^{-\lambda(t_1+t_4)}}\right]e^{-\lambda t_5} \qquad (3)$$

A directional gamma detector may be installed at some point on the external coolant piping, and an indication obtained which is proportional to the power level of the reactor. This indication would have the advantages, over conventional neutron detection systems, of being independent of the control rod configuration within the reactor core; and would use crystalline materials in which there is no apparent permanent damage due to radiation (i.e., no aging).

2. While the experimental data does not support this conclusion as fully as might be desired, it is actually quite encouraging considering that:

(a) Data was taken on an 'as available, not to interfere' basis, and data on characteristics which it would have been desirable to investigate was not available;

(b) The general condition at the reactor system, as a result of previous and current tests, was far below standard; and often the operating conditions imposed were abnormal;

(c) The educational background of the writer, at the time the experimental work was done, was imperfect (to say the least) in this field. As a result, it was not realized that many other points should be considered (e.g., concentration of impurities in coolant, etc.).

3. The dependence of the indication upon the flow rate of the coolant for an existing reactor system can be minimized quite easily by proper physical location of the detector. As an example; the percent variation in the flow correction factor for the experimental system was 15.6 over a flow range of 50-100 percent. Had the distance from the reactor outlet to the detector been only one fourth as much (i.e. 6.5 feet instead of 26 feet) the correction factors would have been

for $F = 1.00$; Correction factor $= 0.947e^{-0.052} = 0.898$;

and for $F = 0.50$; Correction factor $= 0.984e^{-0.104} = 0.886$;

or a percent variation in the correction factor of 1.3 over a flow range of 50-100 percent.

For an initial design, the flow correction factor in equation (3) can be written as

$$\left[\frac{1 - e^{-0.693\,\alpha_1 \frac{T}{T_{1/2}}}}{1 - e^{-0.693\frac{T}{T_{1/2}}}}\right] e^{-0.693\,\alpha_2 \frac{T}{T_{1/2}}}$$

Where T = Time for a complete flow cycle;

$T_{1/2}$ = Half life of the radioactive species of concern;

α_1 = Fraction of T required to traverse the core;

α_2 = Fraction of T required to reach the detector.

Examination of the factor in this form indicates that for minimum effect of flow on the indication:

(a) The ratio of flow cycle time to half life should be as large as possible;

(b) The fraction of flow cycle time spent in the core should approach one, as closely as possible; and

(c) The fraction of flow cycle time to reach the detector should approach zero, as closely as possible.

The major effect of (a) and (b) will be to influence the equilibrium concentration of the activated species in the coolant. From the design point of view, (c) is the most critical; yet the simplest to control.

Based on the design form of the correction factor, it is possible to plot design curves of percent variation in correction factor (effectively percent error in indication) versus the ratio of flow cycle time to half life. A different curve will be obtained for each combination of α_1 and α_2, however, the only combinations which appear to be of any practical interest are those of $0.5 \leq \alpha_1 \leq 1.0$ and $0.0 \leq \alpha_2 \leq 0.5$. Plotting these design curves, which involves a multitude of simple calculations, is considered by the writer to be beyond the scope of this paper. Through the use of the design curves or the design form of the correction factor, it is possible to optimize the position of the detector and the ratio of loop time to core time for any given system, using any desired coolant, so that the maximum percent error in the indication will be within any desired range.

4. There are two effects on the gamma activity in the reactor coolant which are reflections of the chemical elements which make up the coolant and the impurities present.

The first effect is on the decay constant, λ . The λ used in the equation which describes the induced delayed gamma activity in the reactor coolant must be a weighted average λ . As such, it will be relatively strongly influenced by the presence of small amounts of other than normal radioactive substances with λ's which vary considerably from the average for the pure coolant.

The second effect is on the slope of the relationship between the power level and the gamma indication. The slope of this relationship is directly proportional to the macroscopic cross-section of the reactor coolant. Obviously this cross-section will be strongly influenced by small amounts of large cross-section elements, either normally present in the coolant or introduced as impurities.

If the indication of power level is based on the gross gamma activity, as was the case with the experimental system, slight changes in the concentration of impurities present, or the introduction of very small amounts of fission products into the coolant system, can produce marked changes in the slope of the relationship and the effect of flow on the indication. While no experimental data can be produced to support this, as data was not available on specific coolant composition during the tests, it was known that the composition was variable from operating period to operating period. This would account for the variation of slope from run to run in the experimental data, and for some of the seemingly "bad" points on the resultant plot.

The first of these effects can be effectively eliminated by a rather simple device. If only one specific activity were to be observed, and all other gamma activity eliminated from the detection system, the λ of concern would be the λ for the specific activity and would remain constant.

Therefore a simple fixed range pulse height selector placed in the circuit ahead of the indicator would pass only the scintillations produced by the gammas from the selected decay process, and the power level indication would depend only on the selected activity. This also suggests the possibility of introducing small amounts of some tracer element into a coolant which did not have a neutron induced delayed gamma reaction of its own which would be suitable for detection purposes.

The second effect is more difficult to control, however it should be remembered that under normal operating conditions the concentration and number of impurities in a reactor system would be much smaller than in the experimental or prototype system. The control of impurities in water and other liquids is not a new problem to engineers and can often be handled in a relatively simple manner. In a pressurized water system , this could be accomplished by partial continuous or intermittent circulation of the coolant through a mechanical filter, then resin demineralizers. It may be that both cation and anion exchange should be utilized for maximum effectiveness. This would be a system similar to some of those proposed for use in connection with boiling-water reactor systems. As the system would be a closed system, the impurities would not built up rapidly, and continuous checking of the impurity concentration would not be required. Intermittent checks of the impurity concentration could be accomplished by a modification of the low level monitor system developed by E. I. duPont de Nemours and Company (2). Chemical analysis could of course be used, but is somewhat more involved, and requires personnel trained in chemical analysis techniques. With controlled impurity concentration, no slope correction circuit would be required in the metering circuit, but could

be included as an added precaution in the event that an occasional thermal calibration should indicate a slope correction was needed.

5. It is the opinion of the writer that it is quite possible to design a system for power level determination which uses the neutron induced gamma activity in a reactor coolant as the source for the indication. Such a system can be made as accurate as desired, and independent of such things as flow rates, control rod configurations, deterioration of materials (aging), and impurities in the coolant. By suitable choice of certain design variables, the system can be made applicable to any existing or proposed nuclear reactor system which circulates a coolant through a region of neutron flux. The system may be designed for any type of coolant, organic or in-organic; and is equally applicable to boiling reactor systems. In the latter, it appears that the impurity problem is not as serious as in those systems where no phase change (i.e. liquid-vapor) in the coolant is encountered.

No attempt has been made to evaluate the transient behavior of this system, however observations of the experimental system during the experimental period indicated that it does have many desirable features under transient conditions.

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W2225 Wallace
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